ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
OCTOBER, 1996

Graph Paper PRINTED IN U.S.A.

FIGURE 3.1.A-1
PORV OPENING PRESSURE FOR OPERATION LESS THAN OR EQUAL TO 305°F

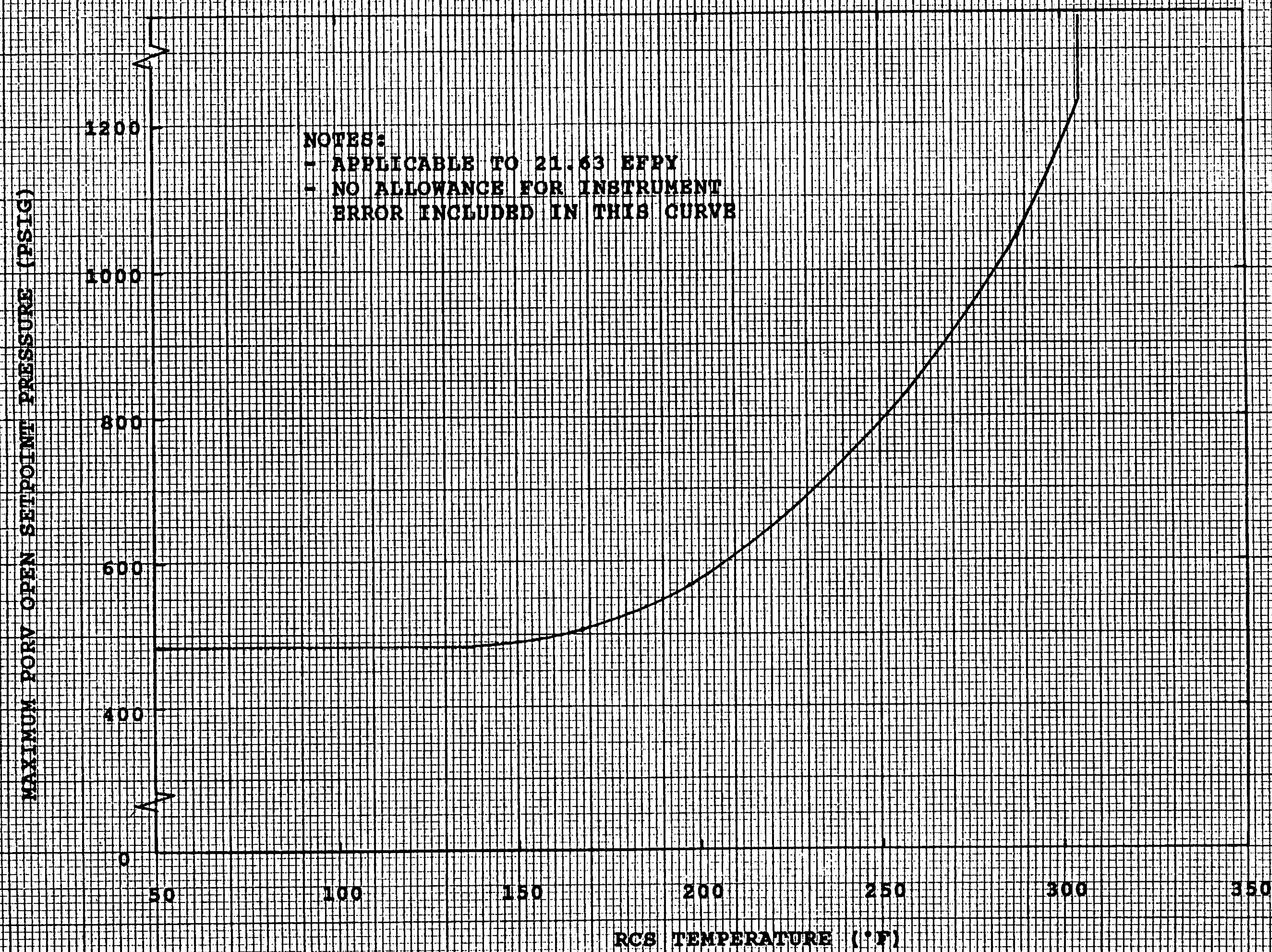


FIGURE 3.1.A-2
MAXIMUM PRESSURIZER LEVEL WITH PORVs
INOPERABLE AND ONE CHARGING PUMP ENERGIZED

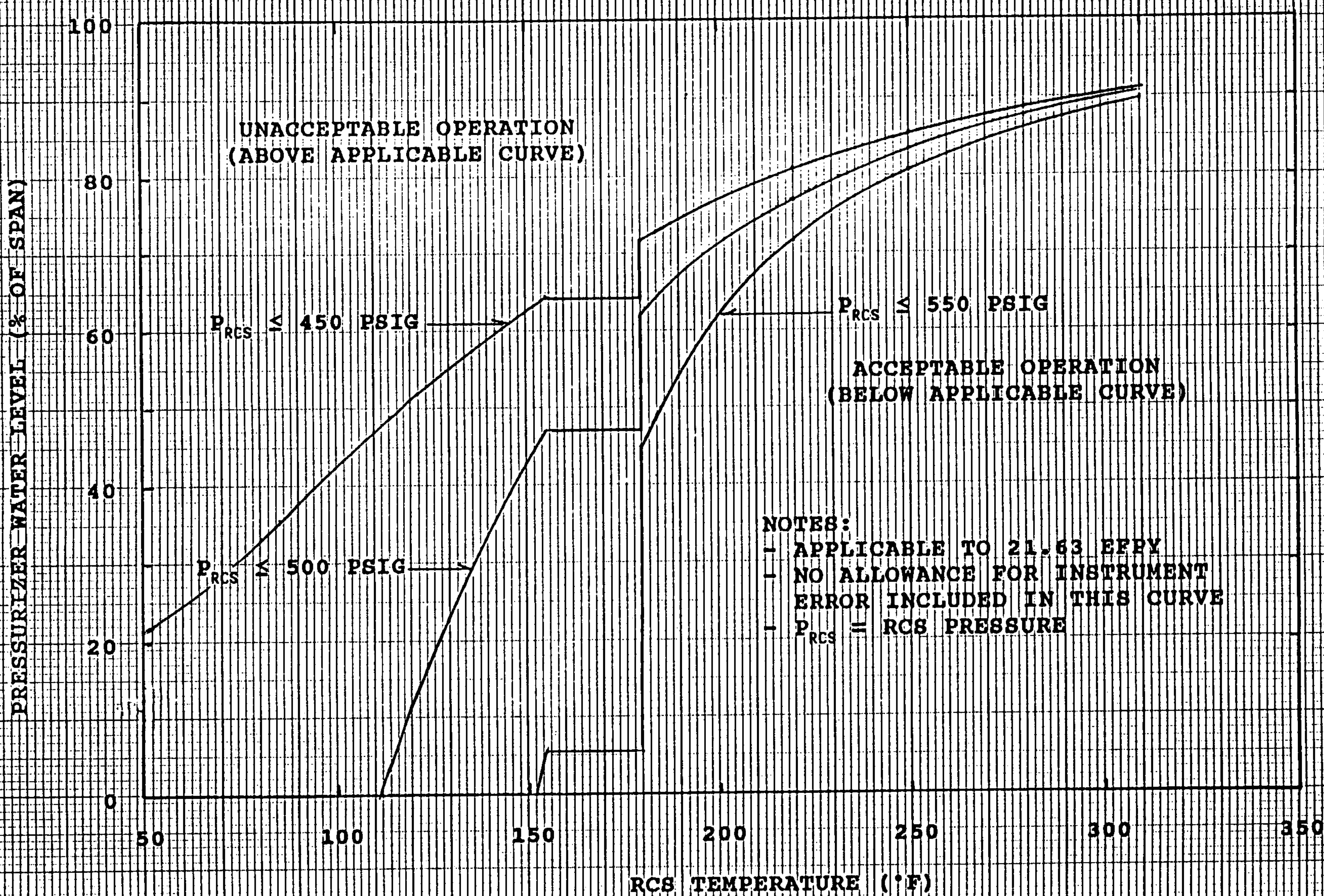
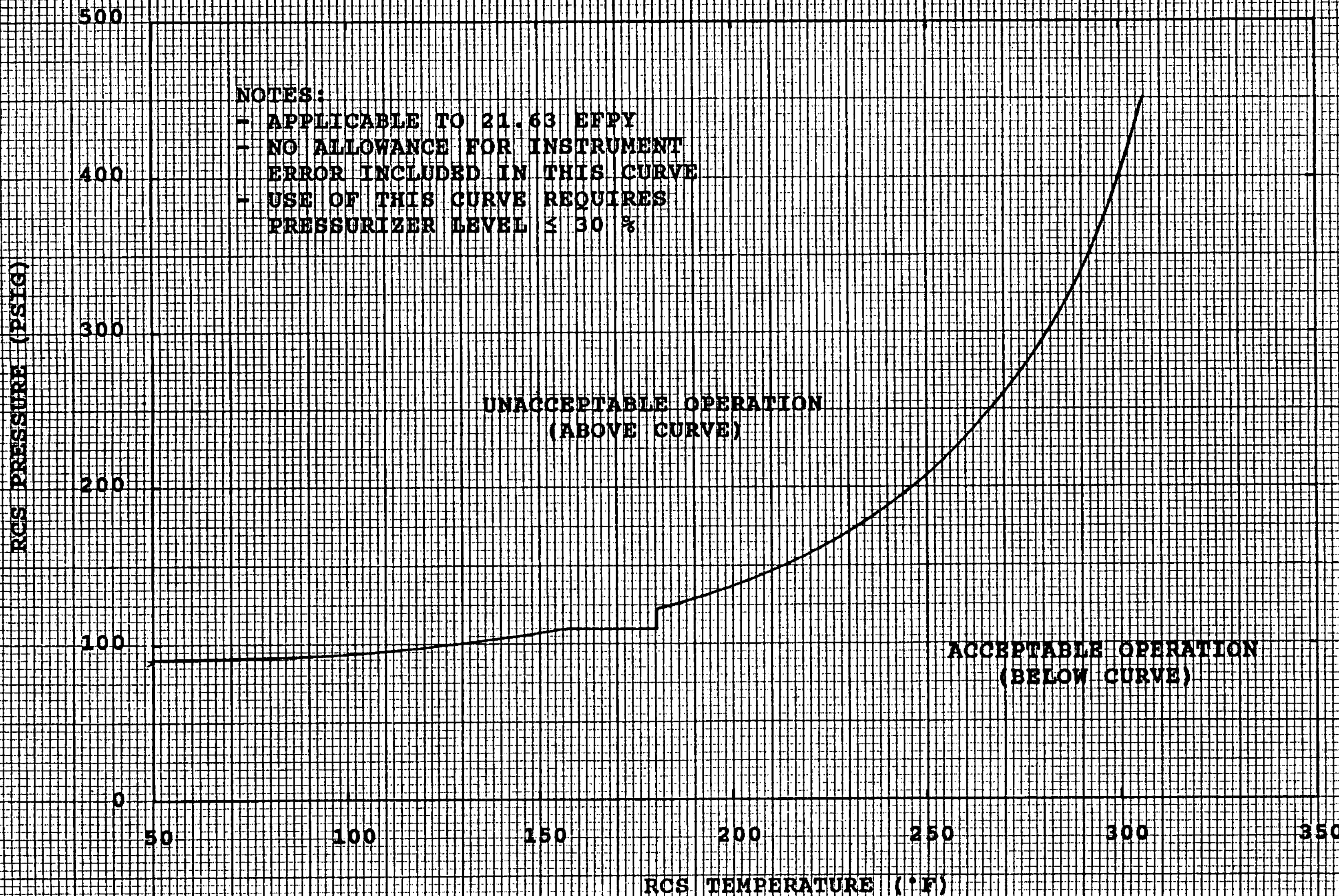


FIGURE 3.1.A-3
MAXIMUM REACTOR COOLANT SYSTEM PRESSURE FOR OPERATION WITH PORVs INOPERABLE AND ONE SAFETY INJECTION PUMP AND/OR THREE CHARGING PUMPs ENERGIZED



B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) averaged over one hour shall be limited in accordance with Figure 3.1.B-1 and Figure 3.1.B-2 for the service period up to 21.63 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those present may be obtained by interpolation.

   b. Figure 3.1.B-1 and Figure 3.1.B-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figure 3.1.B-1 and Figure 3.1.B-2 shall be recalculated periodically using methods discussed in WCAP-7924A and WCAP-12796 and results of surveillance specimen testing as covered in WCAP-7323[7] and as specified in Specification 3.1.B.3 below. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than six months prior to scheduled specimen removal.

3. The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and Charpy V notch (wedge open loading) testing of specimens.

---

* Refer to UFSAR Section 4.5, WCAP-7323, and Indian Point Unit No. 2, "Application for Amendment to Operating License," sworn to on February 3, 1981.

The specimens will be removed and examined at the following intervals:

Capsule 1	End of Cycle 1 operation
Capsule 2	End of Cycle 2 operation
Capsule 3	End of Cycle 5 operation
Capsule 4	End of Cycle 8 operation
Capsule 5	End of Cycle 16 operation
Capsule 6	Spare

4. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

5. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

6. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

Basis

Fracture Toughness Properties

All components in the Reactor Coolant System are designed to withstand the effects of the cyclic loads due to reactor system temperature and pressure changes[(1)]. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the UFSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation[(2)].

The reactor vessel plate opposite the core has been purchased to a specified Charpy V-notch test result of 30 ft-lb or greater at a Nil-Ductility Transition Temperature (NDTT) of 40°F or less. The material has been tested to verify conformity to specified requirements and a NDTT value of 20°F has been determined. In addition, this plate has been 100 percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other Reactor Coolant System components, meet the appropriate design code requirements and specific component function[3].

As a result of fast neutron irradiation in the region of the core, there will be an increase in the Reference Nil-Ductility Transition Temperature (RT_{NDT}) with nuclear operation. The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample location are described in Appendix 4A of the UFSAR. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The current heatup and cooldown curves are based upon a maximum fluence of 0.98×10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operation for a nominal period of 21.63 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level and beyond Cycle 9 for 12 EFPYs at 3071.4 MWt power level and T average of 579.7°F). Any changes in the operating conditions could result in an extension of the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.

The actual shift in RT_{NDT} will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. These samples are evaluated according to ASTM E185[6]. To compensate for any increase in the RT_{NDT} caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4] and WCAP-12796[13] and MSE-REME-0076[14].

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. That capsule was tested by Southwest Research Institute (SWRI) and the results were evaluated and reported[8,9]. The second surveillance capsule was removed during the 1978 refueling outage. That capsule has been tested by SWRI and the results have been evaluated and reported[10]. The third vessel material surveillance capsule was removed during the 1982 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported[11]. The fourth surveillance capsule was removed during the 1987 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported[12]. Heatup and cooldown curves (Figures 3.1.B-1 and 3.1.B-2) were developed by Westinghouse[13]. These curves are essentially identical to those obtained using the new Appendix G methods[14].

The maximum shift in RT_{NDT} at a fluence of 0.98 X 10^{19} n/cm^2, (nominal 21.63 EFPYs of operation) is projected to be 155.5°F at the 1/4 T and 105°F at the 3/4 T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for this plate of the IP2 reactor vessel was 21°F. The heatup and cooldown curves have been computed on the basis of the RT_{NDT} of Plate B2002-3 because it is anticipated that the RT_{NDT} of the reactor vessel beltline material will be highest for Plate B2002-3, at least for the above fluence[12].

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and are discussed in detail in WCAP-7924A[4] and WCAP-12796[13] and MSE-REME-0076[14].

The approach specifies that the allowable total stress intensity factor (K_I), at any time during heatup or cooldown, cannot be greater than that shown on the K_{IR} curve[5] for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (1)$$

where:

K_{Im} is the stress intensity factor caused by membrane (pressure) stress,

K_{It} is the stress intensity factor caused by the thermal gradients,

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state condition (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature, and thus tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure- and temperature-sensing instruments.

The use of the composite curve becomes mandatory in setting heatup limitations because it is possible for conditions to exist such that, over the course of the heatup ramp, the controlling analysis switches from the O.D. to the I.D. location, and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at 1/4 T. The thermal gradients induced during cooldown tend to produce tensile stresses at the 1/4 T location and compressive stresses at the 3/4 T position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel I.D. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1.B-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

(1) Indian Point Unit No. 2 UFSAR, Section 4.1.5.

(2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.

(3) Indian Point Unit No. 2 UFSAR, Section 4.2.5.

(4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.

(5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G. (6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.

(7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, May 1969.

(8) Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

(9) Supplement to Final Report - SWRI Project No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.

(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

(11) Final Report - SWRI Project No. 06-7379 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Z," E.B. Norris, April 1984.

(12) Final Report - SWRI Project No. 17-2108 (Revised)- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V," F.A. Iddings - SWRI, March, 1990.

(13) WCAP-12796, "Heatup and Cooldown Limit Curves for the Consolidated Edison Company Indian Point Unit 2 Reactor Vessel," N.K. Ray, Westinghouse Electric Corporation.

(14) MSE-REME-0076, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P.A. Grendys, Westinghouse Electric Corporation.

FIGURE 3.1.B-1
COOLANT SYSTEM HEATUP LIMITATIONS

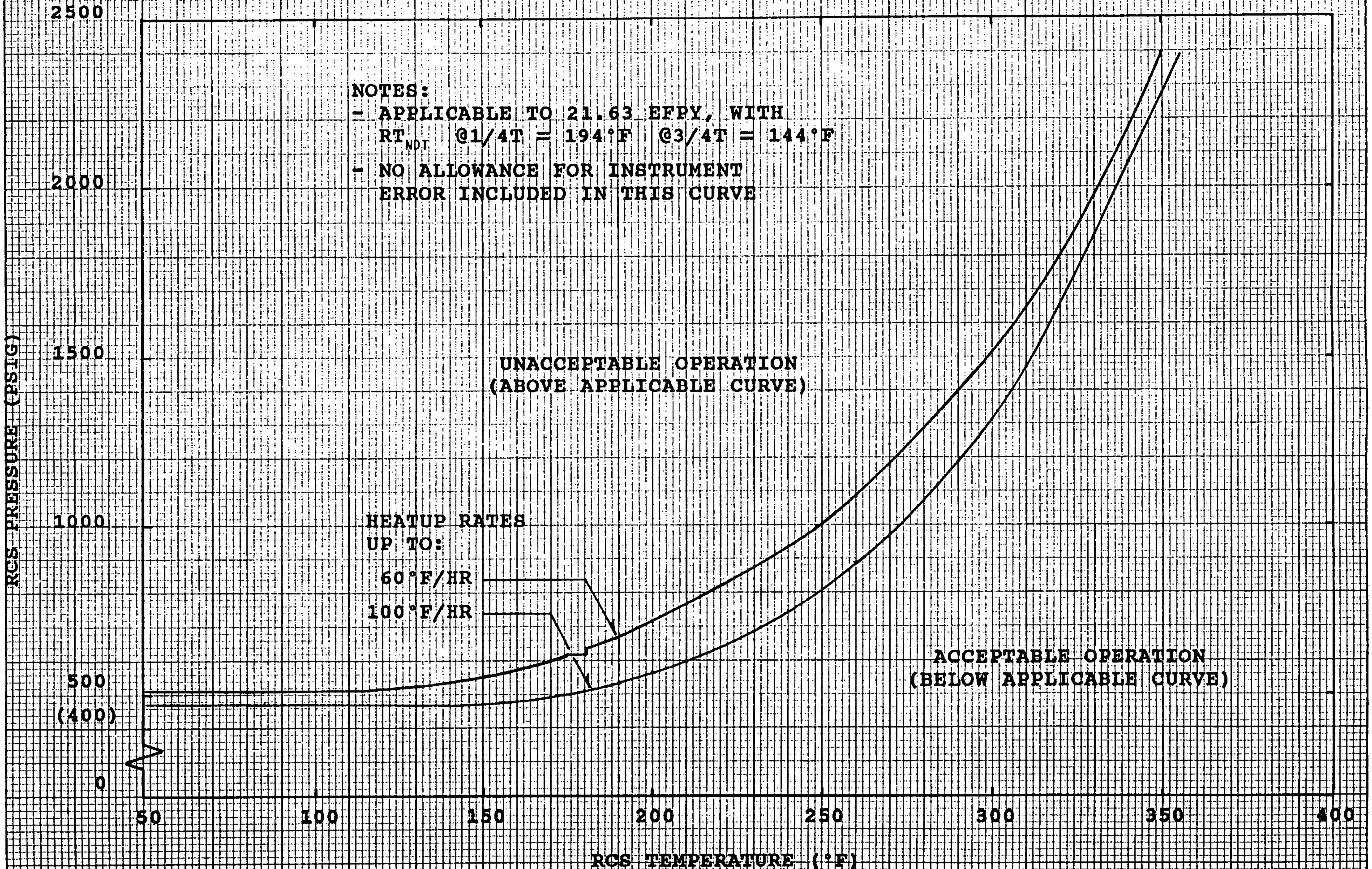
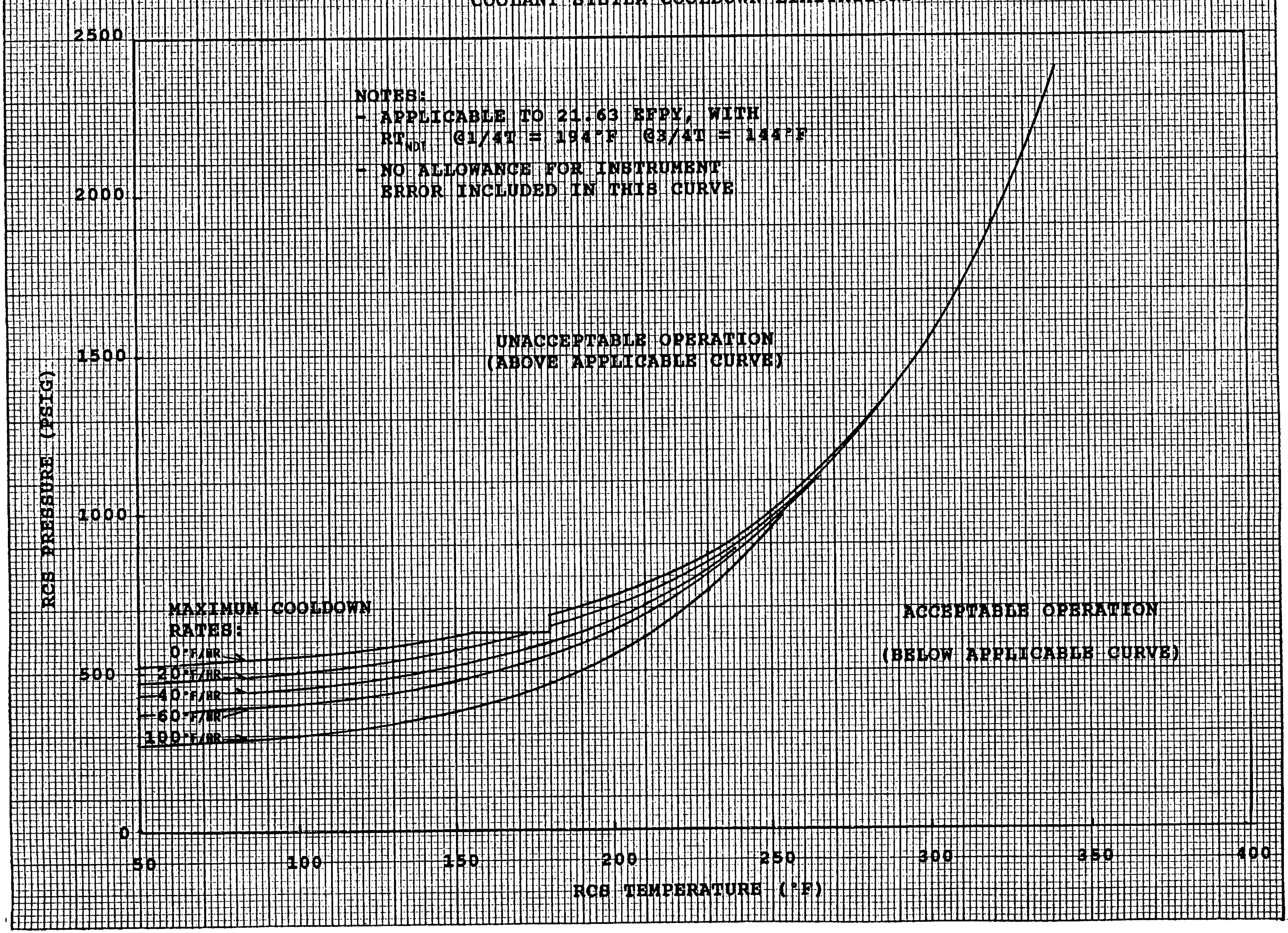


FIGURE 3.1.B-2
COOLANT SYSTEM COOLDOWN LIMITATIONS



4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specifications

a. When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b. When Reactor Coolant System modification or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c. The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 21.63 effective full-power years of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressure during cooldown for the leak test temperature shall be in accordance with Figure 3.1.B-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak-tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G and the methods described in reference 13 of Technical Specification 3.1.B. This code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first 21.63 effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 194°F. The minimum inservice leak test temperature requirements for periods up to 21.63 effective full-power years are shown on Figure 4.3-1.

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1.B-2 must not be exceeded. Figures 4.3-1 and 3.1.B-2 are recalculated periodically, using methods discussed in WCAP-7924A, WCAP-12796 and MSE-REME-0076 and results of surveillance specimen testing, as covered in WCAP-7323.

The current heatup and cooldown curves are based upon a maximum fluence of 0.98×10^{19} n/cm^2 at the inner reactor vessel surface (45° angle, vessel belt line). This fluence is based upon plant operaton for a nominal period of 21.63 EFPYs (Operation up to Cycle 9 for 9.63 EFPYs at 2758 MWt power level and beyond Cycle 9 for 12 EFPYs at 3071.4 MWt power level and T average of 579.7°F). Any changes in the operating conditions could result in an extension of the allowable EFPYs, since the fluence (or ΔRT_{NDT} due to irradiation) is the controlling factor in the generation of these curves.

<u>Reference</u>

UFSAR Section 4

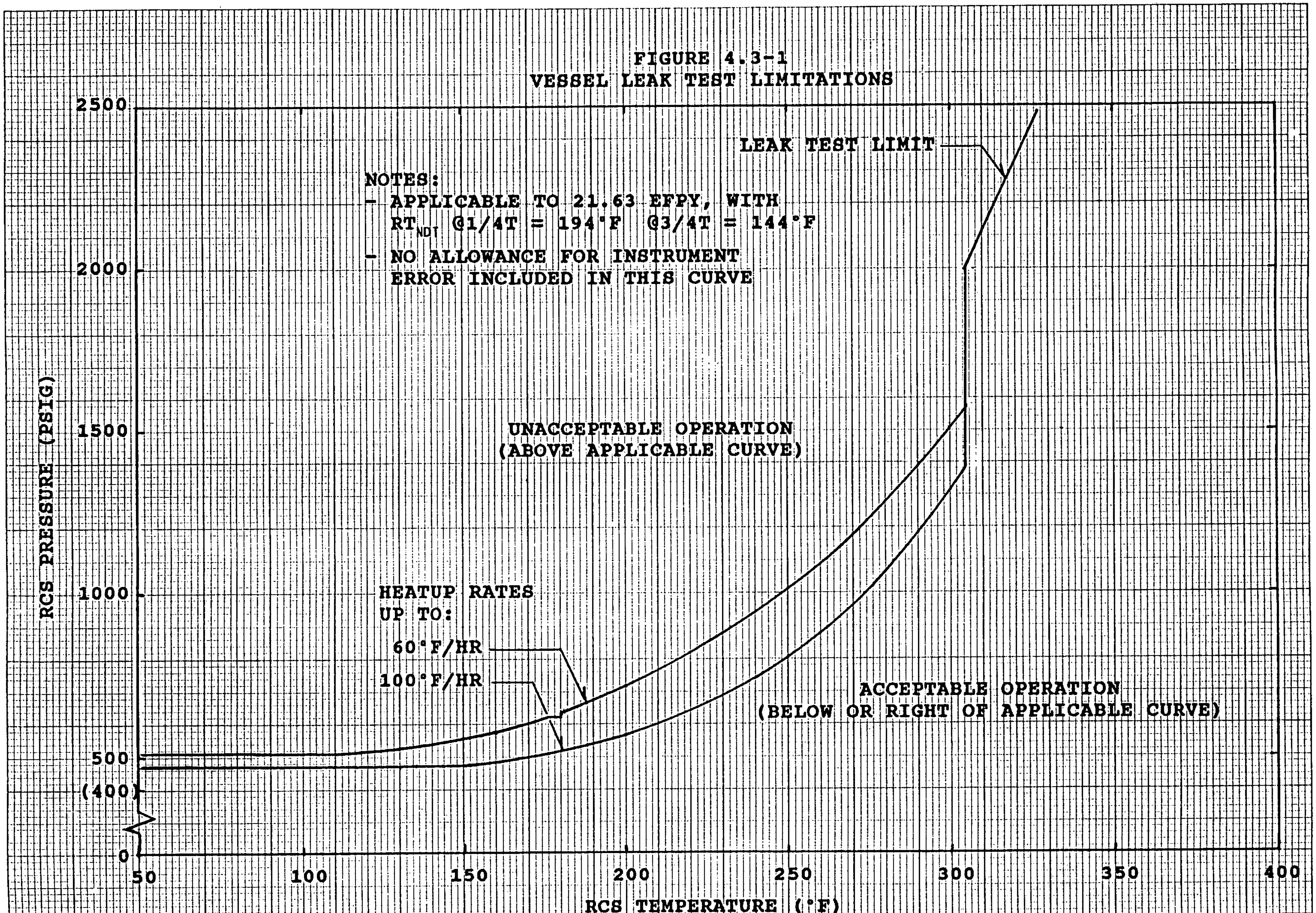


FIGURE 4.3-1
VESSEL LEAK TEST LIMITATIONS
LEAK TEST LIMIT
NOTES:
- APPLICABLE TO 21.63 EFPY, WITH
RT_{NDT} @1/4T = 194°F @3/4T = 144°F
- NO ALLOWANCE FOR INSTRUMENT
ERROR INCLUDED IN THIS CURVE
UNACCEPTABLE OPERATION
(ABOVE APPLICABLE CURVE)
HEATUP RATES
UP TO:
60°F/HR
100°F/HR
ACCEPTABLE OPERATION
(BELOW OR RIGHT OF APPLICABLE CURVE)
RCS PRESSURE (PSIG)
2500
2000
1500
1000
500
(400)
0
50
100
150
200
250
300
350
400
RCS TEMPERATURE (°F)

ATTACHMENT II

SAFETY ASSESSMENT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
OCTOBER, 1996

SECTION I - Description of Change

This application for amendment to the Indian Point 2 Technical Specifications seeks to amend Figures 3.1.A-1, 3.1.A-2, and 3.1.A-3, Section 3.1.B and its bases, Figures 3.1.B-1 and 3.1.B-2and the bases of Section 4.3 and Figure 4.3-1.

The proposed changes are contained in Attachment I to the Application for Amendment enclosed with this letter, while the associated Safety Assessment is provided in Attachment II. Attachment III "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation", MSE-REME-007, describes the methodology used in developing the pressure-temperature curves. Attachment IV, describes the changes to ASME Appendix G as approved by the Section XI Subcommittee, and Attachment V describes the new calculation of stress intensity factors.

On March 7, 1991, Con Edison submitted Technical Specification changes to adapt new Heatup and Cooldown Curves. The curves were generated in accordance with NRC Generic Letter, GL 88-11, NRC Position on Radiation Embrittlements of Reactor Vessel Materials and Its Impact on Plant Operations, July 12, 1988 which directed utilities to use the methodology of Regulatory Guide (RG) 1.99 Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials to produce the effect on neutron radiation on reactor vessel materials. The heatup and cooldown curves were generated using the Raju-Newman method for calculating stress intensity factors.

On October 21, 1991, the NRC issued "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 2 (TAC No. 79910)". Since the Raju-Newman method is not in Standard Review Plan (SRP) 5.3.2 and the Raju-Newman method was then under review for inclusion in Section XI of the ASME Code, the NRC Staff, considered that its acceptance was premature in allowing the use of the Raju-Newman method in the P/T limits calculations. Nevertheless, the staff determined that the proposed limits would be applicable for limited EFPY for the proposed P/T limits. The 60°F/hr. curves were approved up to 16 EFPYs and 100°F/hr curves to 12 EFPYs.

Currently, Indian Point 2 is at approximately 13 EFPYs, the 100°F/hr curves cannot be used and the plant is fast approaching 16 EFPYs.

By letters dated July 6, 1992 and October 12, 1993, Con Edison provided its response to GL 92-02 Rev. 1. On April 20, 1994, the NRC determined that Con Edison had provided all necessary information for GL 92-01. The NRC provided Con Edison with its reactor vessel information database (RVID) for comment. Con Edison provided its comments on the RVID in a letter dated July 12, 1994.

By letters dated August 17, 1995 and November 20, 1995, Con Edison responded to Generic Letter GL 92-01 Rev. 1, Supplement 1.

The above responses have shown that IP2 meets all criteria for Reactor Vessel integrity for at least the remainder of its licensing period.

The ASME operating plant criteria working group has given its approval to a "Raju-Newman" type method. The Evaluations Subgroup, the Section XI Subcommittee, the Boiler and Pressure Vessel Main Committee and the ASME Board of Nuclear Codes and Standards have also approved it.

SECTION II - Evaluation of Changes

We have compared the Indian Point 2 21.63 EFPY heatup/cooldown curves versus those calculated using the method adopted by the Code Committees and they are within 2% of each other. The differences cannot be discerned from the plotted curves. Therefore we request the removal of the limiting conditions as noted by asterisks in our current specifications. In addition, for clarification, the graphs have been extended to lower temperatures. Table of values will be provided to the operators.

SECTION III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed Application involves no significant hazards based on the following information:

1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

Neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed changes. The adjusted reference temperature of the most limiting beltline material was used to correct the pressure-temperature (P-T) curves to account for irradiation effects. Thus, the operating limits are adjusted to incorporate both the initial fracture toughness conservatism present when the reactor vessel was new and the effect of fluence. The adjusted reference temperature calculations were performed utilizing the guidance contained in RG 1.99, Revision 2. Overpressure Protection System (OPS) curves and tables were regenerated to be consistent with the new P-T curves. The updated curves provide assurance that brittle fracture of the reactor vessel is prevented.

2) Does the proposed license amendment create the possibility of a new or different kind of accident from any previously evaluated?

Response:

The updated P-T and OPS limits will not create the possibility of a new or different kind of accident. The revised operating limits merely update the existing limits by taking into account the effects of radiation embrittlement, utilizing criteria defined in RG 1.99, Revision 2. The updated curves are conservatively adjusted to account for the effect of irradiation on the limiting reactor vessel material.

No change is being made to the way the pressure-temperature limits provide plant protection. No new modes of operation are involved. Incorporating this amendment does not necessitate physical alteration of the plant.

3) Does the proposed amendment involve a significant reduction in the margin of safety?

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The pressure-temperature operating limits and OPS setpoints are designed to maintain an appropriate margin of safety. The required margin is specified in ASME Boiler and Pressure Vessel Code, Section III, Appendix G and 10 CFR 50 Appendix G. The revised curves are based on the latest NRC guidelines along with actual neutron fluence data for the reactor vessel. The new limits retain a margin of safety equivalent to the original margin when the vessel was new and the fracture toughness was slightly greater. The new operating limits account for irradiation embrittlement effects, thereby maintaining a conservative margin of safety.

The removal of the pressure-temperature limits for criticality does not reduce the plant safety margin because these limits are conservatively encompassed and bounded by the requirements of the proposed Technical Specification 3.1.C.2.

SECTION IV - Impact of Changes

These changes will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

SECTION V - Conclusions

The incorporation of these changes: a) will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a new or different kind from any evaluated previously in the safety analysis report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC). Both Committees concur that the proposed changes do not represent a significant hazards consideration.

ATTACHMENT III

INDIAN POINT UNIT 2 HEATUP AND COOLDOWN LIMIT
CURVES FOR NORMAL OPERATION, MSE-REME-0076

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
OCTOBER, 1996